Exelon Generation.

10CFR 50.59, 10CFR 72.48, NEI 99-04 (SECY 00-0045)

January 28, 2013

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, DC 20555-0001

Peach Bottom Atomic Power Station (PBAPS), Units 1, 2 and 3 and
PBAPS Independent Spent Fuel Storage Installation (ISFSI)
Facility Operating License No. DPR-12
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-171, 50-277, 50-278, and 72-29 (ISFSI)

Subject: Biennial 10CFR 50.59 and 10CFR 72.48 Reports for the Period 1/1/2011 through 12/31/2012 and Annual Commitment Revision Report for the Period 1/1/12 through 12/31/12

Enclosed are the 2010-2011 Biennial 10CFR 50.59 and 10CFR 72.48 Reports and the 2012 Annual Commitment Revision Report as required by 10CFR 50.59(d)(2), 10CFR 72.48, and SECY-00-0045 (NEI 99-04).

There are no new regulatory commitments contained in this transmittal.

If you have any questions or require additional information, please contact D. J. Foss at 717-456-4311.

Sincerely,

Patrick D. Navin
Plant Manager
Peach Bottom Atomic Power Station

cc: Senior Resident Inspector, USNRC, PBAPS
Commonwealth of Pennsylvania
Document Control Desk, USNRC, Washington DC

CCN: 13-01

Attachments

WMSS26
IE47 / FSME20
NRR
FSME

Exelon Nuclear
Peach Bottom Atomic Power Station

Docket Nos. 50-171
50-277
50-278
72-29

2011-2012 Biennial 10CFR 50.59 and 10CFR 72.48 Reports and the 2012 Commitment Revision Report

These reports are issued pursuant to reporting requirements for Peach Bottom Atomic Power Station Units 1, 2 and 3. These reports address tests and changes to the facility and procedures as they are described in the Peach Bottom Final Safety Analysis Report and Independent Fuel Storage Safety Analysis Report for the TN-68 Spent Fuel Cask. These reports consist of those tests and changes that were implemented between January 1, 2011 and December 31, 2012. Also, this report identifies commitments that were revised during 2012 and require reporting in accordance with the guidelines of NEI 99-04, Managing Regulatory Commitments Made By Power Reactor Licensees to the NRC Staff endorsed by SECY-00-0045.

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Exelon Nuclear
Peach Bottom Atomic Power Station
Units 1, 2 and 3
Docket Nos. 50-171, 50-277, and 50-278

BIENNIAL 10CFR 50.59 REPORT
JANUARY 1, 2011 THROUGH DECEMBER 31, 2012
EVALUATION SUMMARIES

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Title: Replacement of Unit 3 Low Pressure Turbine Rotors and Casings

Units Affected: 3

Year Implemented: 2011

Brief Description:

This activity involved installation of new Unit 3 Low Pressure Turbine (LPT) rotors and inner casings. The new turbine rotor will be resistant to Stress Corrosion Cracking (SCC) and the new inner casings will be significantly less susceptible to erosion. The turbine will have improved thermal efficiency. The turbine retrofit will result in increased main generator power output under most operating conditions with no increase in reactor thermal power and no increase in main steam flow rate. The turbine retrofit will also result in changes to power cycle parameters, such as temperature, flow rate, pressure, and moisture content. Operation of the plant under these different conditions was evaluated by this activity. The activity will not change the UFSAR design function of the turbine, condenser or any connected system.

Summary of Evaluation:

Reactor Thermal Power and HP turbine steam path remain the same; therefore, no impacts have been identified on the main steam bypass system. The system will operate as described in the UFSAR. The activity will result in a small decrease in the nominal final feed water temperature with the new LP Turbines. Final feed water temperature will be slightly lower following LP turbine retrofit. The reload licensing transient analyses described in UFSAR section 14.5.2.3 assumes an initial Final feed water Temperature of 381.5 Degrees F and are insensitive to variations of +/- 10 Degrees F. Since final feed water Temperature will change from approximately 378 Degrees F to approximately 375.5 Degrees F, these evaluations remain valid. Heat removal duty on feed water heaters will decrease slightly. This will not adversely affect transients evaluated in the UFSAR (e.g., loss of feed water heating). It also does not have an adverse affect on the fuel analysis. Use of the vendor methodology for determining turbine missile probability and changes to input parameters for that analysis to reflect the retrofit turbine design constitute acceptable changes to an approved methodology.

There are no changes required to the Operating License or Technical Specifications required by this activity.

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Title: Early Hydrogen Water Chemistry Demonstration

Units Affected: 3

Year Implemented: 2011

Brief Description:

As part of an Electric Power Research Institute (EPRI) project, Exelon performed an Early Hydrogen Water Chemistry (EHWC) demonstration at Peach Bottom Unit 3. Hydrogen injection during heat-up and startup conditions reduces the potential for IGSCC (Intergranular Stress Corrosion Cracking) and will extend the useful life of reactor internals and piping. Temporary equipment was installed for the purpose of injecting hydrogen into the Reactor Recirculation System and the Feedwater System during startup at reactor coolant temperatures below 200°F and up to reactor power of approximately 15%. The existing HWC system is started at 13 to 15% power. This temporary equipment was: (a) installed during an outage, (b) used during startup, (c) isolated after use, (d) purged of hydrogen, and (e) disconnected and stored when the demonstration was completed. Plant systems and connection points were restored to the original configuration.

Summary of Evaluation:

The purpose was to demonstrate that sufficient hydrogen gas can be injected into the reactor coolant through the Reactor Recirculation system and Feedwater system during startup so that IGSCC (Intergranular Stress Corrosion Cracking) is mitigated. This was the first United States demonstration of EHWC, but the process has been previously demonstrated in Japan at Shimane 2 and Tokai 2. However, Peach Bottom 3 is a noble metals plant whereas Shimane 2 and Tokai 2 are not noble metals plants. Therefore, this will be the first demonstration of EHWC at a noble metals plant. At a noble metals plant, less hydrogen is required to achieve the target molar ratio of hydrogen to oxygen. Reducing the potential for IGSCC by reducing ECP during startup will extend the life of reactor internals and piping. The 50.59 Evaluation determined that this temporary change did not increase the frequency or consequences of a previously evaluated accident or create the possibility of a new accident since no accident initiators are involved. It does not increase the likelihood of occurrence of a previously evaluated malfunction of an SSC important to safety because the affected equipment does not interfere with any previously evaluated. It does not increase the consequences of a previously evaluated malfunction of equipment important to safety because there are no consequences associated with the activity. It does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR because no new failure modes are introduced. It does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered because no system parameters will change as a result of this activity.

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Title: Drywell Equipment Drain Sump Temporary Logic Reconfiguration

Units Affected: 3

Year Implemented: 2012

Brief Description:

As a result of the failure of the 3BP098 Drywell Drain Sump Pump, this activity will temporarily re-configure the sump controls to ensure 3AP098 will always run upon a HI-HI level signal. Instead of having an alternating scheme for the pumps, one pump will perform all pump outs on the HI-Hi level set point. The Hi-Hi alarm will only come on after a time delay if the single pump cannot adequately reduce sump level within the duration allotted by the time delay.

Summary of Evaluation:

The 3AP038 pump will run more frequently, but cycling on the HI-HI set point will minimize the number of pump starts. The Drywell Equipment Drain Pumps and associated logic support Technical Specification (TS) required Reactor Coolant Leakage (RCL) operational leakage determination, but do not perform or support any safety related function. No safety analyses are impacted by this activity. This change does not interfere with the flow monitoring of the Drywell Equipment drains and does not affect the primary containment isolations for the sump drains. This temporary change does not increase the frequency or consequences of a previously evaluated accident or create the possibility of a new accident since no accident initiators are involved. It does not increase the likelihood of occurrence of a previously evaluated malfunction of an SSC important to safety because the affected equipment does not interfere with any previously evaluated. It does not increase the consequences of a previously evaluated malfunction of equipment important to safety because there are no consequences associated with the drywell sump pumps. It does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR because no new failure modes are introduced. It does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered because no system parameters will change as a result of this activity.

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Title: Compensatory Actions for Operability Evaluation for the Unit 2 New Fuel Receipt Evolution in the Spent Fuel Pool

Units Affected: 2

Year Implemented: 2012

Brief Description:

This activity involves the application of procedural controls in order to temporarily consider approximately 156 PB2 degraded Spent Fuel Pool (SFP) cells operable during refueling outage P2R19 based on the results of Operability Evaluation 10-007, Rev 3. These procedural controls ensured that on a temporary basis, the fuel loaded into the SFP cells previously considered to be inoperable will be limited to P2R19 new unirradiated GNF2 fuel with a known reactivity value that has been evaluated to be acceptable in the Operability Evaluation. Technical Evaluation 1380305-02 and Operability Evaluation 10-007, Rev. 3 provide the technical justification for placing P2R19 new unirradiated GNF2 fuel in previously declared inoperable PB2 rack cells due to Boraflex degradation.

Summary of Evaluation:

Administrative controls currently established by existing fuel move sheet development procedure and fuel move tracking software used to generate fuel move sheets assured that only the direction to move new unirradiated fuel into the inoperable cells is provided to Reactor Services. The administrative controls currently established in fuel handling procedures assure that the fuel movement specified in the fuel move sheet is followed and that only new unirradiated fuel is placed in the inoperable cells. This temporary change does not increase the frequency or consequences of a previously evaluated accident or create the possibility of a new accident. It does not increase the likelihood of occurrence of a previously evaluated malfunction of an SSC important to safety. It does not increase the consequences of a previously evaluated malfunction of equipment important to safety. It does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR because no new failure modes are introduced. It does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered because no system parameters will change as a result of this activity.

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Title: Replacement of Unit 2 Low Pressure Turbine Rotors and Casings

Units Affected: 2

Year Implemented: 2012

Brief Description:

This activity involved installation of new Unit 2 Low Pressure Turbine (LPT) rotors and inner casings. The new turbine rotor will be resistant to Stress Corrosion Cracking (SCC) and the new inner casings will be significantly less susceptible to erosion. The turbine will have improved thermal efficiency. The turbine retrofit will result in increased main generator power output under most operating conditions with no increase in reactor thermal power and no increase in main steam flow rate. The turbine retrofit will also result in changes to power cycle parameters, such as temperature, flow rate, pressure, and moisture content. Operation of the plant under these different conditions was evaluated by this activity. The activity will not change the UFSAR design function of the turbine, condenser or any connected system.

Summary of Evaluation:

Reactor Thermal Power and HP turbine steam path remain the same; therefore, no impacts have been identified on the main steam bypass system. The system will operate as described in the UFSAR. The activity will result in a small decrease in the nominal final feed water temperature with the new LP Turbines. Final feed water temperature will be slightly lower following LP turbine retrofit. The reload licensing transient analyses described in UFSAR section 14.5.2.3 assumes an initial Final feed water Temperature of 381.5 Degrees F and are insensitive to variations of +/- 10 Degrees F. Since final feed water Temperature will change from approximately 378 Degrees F to approximately 375.5 Degrees F, these evaluations remain valid. Heat removal duty on feed water heaters will decrease slightly. This will not adversely affect transients evaluated in the UFSAR (e.g., loss of feed water heating). It also does not have an adverse affect on the fuel analysis. Use of the vendor methodology for determining turbine missile probability and changes to input parameters for that analysis to reflect the retrofit turbine design constitute acceptable changes to an approved methodology. There are no changes required to the Operating License or Technical Specifications required by this activity.

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Title: Application of TRACG04 Version 4.2.69.0 for OPRM Set Point Determination

Units Affected: 2 / 3

Year Implemented: 2012

Brief Description:

This activity addresses the use of the General Electric Hitachi (GEH) advanced, multi-purpose NSSS thermal-hydraulic transient code TRACG04P, Version 4.2.69.0, for the purpose of determining the Oscillation Power Range Monitor (OPRM) set points for Peach Bottom Atomic Power Station. OPRM set points are determined for each operating cycle as part of the standard reload licensing process performed in accordance with General Electric's Standard Application for Reactor Fuel (GESTAR II) methodology. The cycle specific OPRM set points are presented in the Core Operating Limits Report (COLR). Version 4.2.69.0 of TRACG04P is an upgraded version of the NRC approved TRACG02A program originally developed and licensed to determine OPRM set points. Version 4.2.69.0 of TRACG04P has not been generically approved by the NRC for OPRM set point determination. OPRM system trip functions are described in the UFSAR and the evaluation of OPRM PBDA set points is performed as part of the Peach Bottom cycle specific safety analysis process. NEDO-32465-A is cited in Technical Specification 3.3.1.1 by reference. Use of TRACG04P Version 4.2.69.0 constitutes a change in methodology. The TRACG02A version of the TRACG thermal-hydraulic code was approved by the NRC and used in the preparation of NEDO-32465-A during the original design and licensing of the GE OPRM system. The TRACG04P code has recently been revised by the vendor, GE-Hitachi (GEH), to address a number of programming issues identified since its initial release.

Summary of Evaluation:

The method of determining OPRM set points is described in the Peach Bottom Technical Specification BASES B.3.3.1.1.2.f. The OPRM trip set points, are established in accordance with approved methodologies and are documented in the Core Operating Limits Report (COLR). The TRACG thermal-hydraulic code supports the determination of the second set of set points, period based detection algorithm (PBDA) trip set points. The TRACG thermal-hydraulic code is used to develop a conservative relationship between the change in fuel bundle critical power ratio (CPR) and the hot bundle oscillation magnitude. This conservative relationship is used to determine the Delta CPR Over Initial MCPR Verses Oscillation Magnitude (DIVOM) curve. The DIVOM curve, in conjunction with the initial maximum critical power ratio (IMCPR) and the hot bundle oscillation magnitude, is used by Global Nuclear Fuels (GNF) to determine the OPRM PBDA set points. The algorithms used to detect thermal-hydraulic instability related neutron flux oscillations, described in Technical Specification BASES B3.3.1.1, are not impacted by this activity. TRACG04P is only used in the set point determination. The slope of the DIVOM curve represents the thermal-hydraulic responsiveness of the fuel to a given oscillation magnitude. Thus, a steeper slope is more conservative than a flatter slope. Benchmarking of the NRC-approved TRACG02A code and the TRACG04P Version 4.2.69.0 code has determined that the DIVOM slope developed using TRACG04P generates a slightly more conservative (steeper) DIVOM slope.

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Title:	Frequency Change to Technical Requirements Manual (TRM) Test Requirement (TR) 3.15 for Alternate Rod Insertion (ARI) Instrumentation
Units Affected:	2 / 3
Year Implemented:	2012

Brief Description:

This activity involved changing the frequency of performance of TR 3.1.5 under the Technical Requirements Manual Specification 3.1 for Alternate Rod Insertion (ARI) Instrumentation. TR 3.1.5 currently requires the performance of a Logic System Functional Test, including scram air header venting and manual actuation logic, once per 24 months. It was proposed that the frequency of testing be reduced from once per 24 months to once per 48 months.

The surveillance interval was changed from 24 months to 48 months for the similar surveillance requirement in Technical Specifications (TS) SR 3.3.4.1.4 involving the Anticipated Transient Without Scram – Recirculation Pump Trip (ATWS-RPT) function in accordance with the Surveillance Frequency Change Program (SFCP). SR 3.3.4.1.4 and TR 3.1.5 are both satisfied by the performance of the same Surveillance Test. Therefore, it is desirable to change the Frequency requirement for TR 3.1.5 to match that of SR 3.3.4.1.4.

Summary of Evaluation:

The Alternate Rod Insertion (ARI) System is specifically listed as a Special Safety System in UFSAR section 1.6.3.4. This activity only affects the frequency of testing the ARI system and does not affect plant operations, the system design basis, or any safety analyses described in the UFSAR. Neither the scope nor the frequency of testing the ARI system is discussed. The function of ARI as described in the UFSAR is to mitigate an ATWS event. The ARI system provides an alternate means of reactor shutdown which is independent of RPS. An ARI signal opens solenoid valves on the scram air header to bleed air from the header which in turn allows the scram inlet and discharge valves to open. An automatic ARI initiation signal takes place simultaneously with Anticipated Transient Without Scram – Recirculation Pump Trip (ATWS-RPT). The venting of the scram header drives the control rods into the core to shut down the reactor. Venting of the scram air header also closes the scram discharge volume vent and drain valves. If automatic or manual insertion of rods fails, the operator injects boron into the reactor using the standby liquid control system (SLCS). Analysis from the Engineering evaluation, quantitative and qualitative internal event risk analyses, review of the UFSAR described functions, and Engineering judgment supports that this frequency change does not have a significant adverse impact on ARI reliably performing its design functions. This change does not increase the frequency or consequences of a previously evaluated accident or create the possibility of a new accident since no accident initiators are involved. It does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR because no new failure modes are introduced. It does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered because no system parameters will change as a result of this activity.

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Title: Correction of Error Identified in Alternate Source Term Calculation

Units Affected: 2 / 3

Year Implemented: 2012

Brief Description:

The activity involves correcting an error made in calculation PM-1057, "Reanalysis of Control Rod Drop Accident Using Alternative Source Terms". During calculation updates for the Operating Cycle PB2C20 core fuel reload, it was discovered that the incorrect Nuclide Inventory File (NIF) was used for the calculation. The NIF provides the RADTRAD program with the core inventory information necessary to calculate the dose consequences at the Control Room, Exclusion Area Boundary (EAB), and Low Population Zone (LPZ). This Activity reanalyzes the core inventory using the correct NIF, which results in an increased dose at the EAB and LPZ. The dose at the control room did not change (to 3 decimal places). UFSAR Table 14.9.7 shows the design basis accident radiological doses at the EAB and LPZ, and therefore needs to be revised to incorporate the corrected doses from PM-1057 for the Control Rod Drop Accident.

Summary of Evaluation:

The dose consequences as given in PM-1057 and UFSAR Table 14.9.7 were revised. There is a minimal increase in dose at the EAB and LPZ locations. There is no increase in the Control Room dose, to three decimal places. The proposed activity, correcting the error contained in PM-1057, results in an adverse change because the dose consequences are increased. Therefore, a 50.59 Evaluation was required. However, the increase in dose is minimal. The increase in dose is well under the less than minimal 10% criteria. In addition, the corrected dose at all locations is less than standard Review Plan limits.

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There were no 10CFR 50.59 Evaluation Reports performed / implemented for Unit 1 during this reporting period.

End of 10CFR 50.59 Report

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Exelon Nuclear
Peach Bottom Atomic Power Station
Independent Spent Fuel Storage Installation (ISFSI)
Docket No. 72-29

BIENNIAL 10CFR 72.48 REPORT
JANUARY 1, 2011 THROUGH DECEMBER 31, 2012
EVALUATION SUMMARIES

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Title: TN-68 Cask Lid Pre-heat and Rework of the Seal Weld

Units Affected: ISFSI Cask TN-68-50

Year Implemented: 2011

Brief Description:

On 9/4/10, an ISFSI Cask Pressure Low alarm was received. In accordance with ISFSI Technical Specification 3.1.5, pressure in the cask overpressure system is required to be maintained above 3.0 atmospheres absolute. The condition was determined to be a result of a manufacturing weld defect found on the lid of ISFSI Cask #50. This defect is allowing helium leakage from the overpressure system. The leak rate is greater than the Technical Specification allowable limit. Spent fuel casks are treated as ASME pressure vessels, limiting repair methods to welding. Since welding was performed on a loaded cask, this activity is considered to be outside the bounds of what the NRC previously had reviewed in the cask SAR.

Summary of Evaluation:

The primary function of the cask metallic seal is confinement. As described in theTN-68 Safety Analysis Report (SAR) Section 4.1, the maximum temperature limit of 536°F (280°C) satisfies the leak tightness function of the metallic seals. The confinement vessel of the cask is analyzed for the design pressure of 100 psig as stated in the UFSAR, Section 3.4.4.1. Calculation 10814-003, Rev. 0 determined the maximum seal temperature as 277°F during the activity, which remains well below the maximum temperature limit for the entire period of rework activities. Calculation 10814-003, Rev. 0 showed that the maximum difference in the average temperatures between the cask lid and the flange remain within 10°F at all times during the rework process. The maximum difference in the average temperatures between the cask lid and the cask flange occur at the beginning of the rework process. Therefore, any relative movement of the cask lid and the cask flange and the subsequent shear stresses on the lid seal are bounded by those at the beginning of the rework process. Further, the bolt loads remain unchanged during the rework activities. The maximum lid seal temperature fluctuation/thermal cycling during the rework process and the relative movement of the cask lid/cask flange and the bolt loads are bounded by the conditions described in the SAR. Therefore, the function of the seals remains unaffected and the integrity of the seals was assured during rework activities. There was no increase in the possibility of a release of radioactive material. There was no affect on cask confinement or structural and shielding functions of the cask. The ability of the cask and other structures, systems and components important to safety to perform their intended safety functions remains unaffected.

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Title: TN-68 Cask Lid Outer Plate and Lid Shield Plate Thickness

Units Affected: ISFSI Casks: TN-68-45-A, TN-68-46-A, TN-68-47-A, TN-68-48-A, TN-68-49-A, TN-68-50-A, TN-68-51-A, TN-68-52-A, TN-68-53-A, TN-68-54-A (and other lids in fabrication)

Year Implemented: 2012

Brief Description:

The activity involved acceptance of TN-68 cask lid outer plates that are locally under thickness. These are plates that were machined on one side subsequent to being distorted (bowed) as a result of welding. The result was a gradual and uniform reduction in thickness from full thickness around the perimeter to an up to 0.12" under thickness condition in the center. The activity approved acceptance of the TN-68 lid shield plates that are locally over thickness up to 0.046".

Summary of Evaluation:

As described in the Safety Analysis Report (SAR), the lid outer plate is part of the confinement vessel and functions as a mounting platform for the top neutron shield and lid plate. There is no increase in the possibility of a release of radioactive material, the affected casks do not contain a defect which could create a substantial safety hazard, and the condition was not reportable under 10CFR21. The cask lid under thickness does not affect the confinement, structural and shielding functions of the cask lid. The ability of the cask lid and other structures, systems and components important to safety to perform their intended safety functions remains unaffected.

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Title: TN-68 Cask Lid Bolts Under-Torque Values

Units Affected: ISFSI TN-68 Casks

Year Implemented: 2012

Brief Description:

A survey of installed lid bolt torques on TN-68 casks at Peach Bottom revealed several bolts had less than required 840-940 ft-lbs torque. A minimum of 300 ft-lbs was measured. This activity evaluated the effect of the reduced torque values on the sealing capabilities of the lid O-rings by revising the associated vendor calculation to include the lower torque value. The calculation revision addressed the ability of the O-rings to maintain their sealing capability during a tip over accident with as little as 300 ft-lbs of torque.

Summary of Evaluation:

The TN-68 Safety Analysis Report (SAR) states that the compressive force always exceeds the tip over force. Since the compressive load on the torque bolts (at 300 ft-lbs) is now less than tip over accident case load, the seal will no longer remain in a compressed state. Therefore, this condition required a 72.48 evaluation. The reanalysis of the lid bolt and O-ring seal does not change any design functions. Therefore, there is no adverse affect on any design function of any cask component due to the reanalysis. The evaluation performed for reduced bolt torque shoed that O-ring sealing capability is maintained and there would be no leakage through the TN-68 O-ring seals with the reduced torque. The reanalysis demonstrated that in the tip over event, a compressive load is maintained on the O-ring seal. The self-energizing design of the O-ring seals allows up to 0.013" of decompression before its sealing capacity is compromised. The reanalysis shows that the tip over event results in a decompression of approximately 0.0043". Based on this analysis, no adverse effect on the confinement function is expected. There was no increase in the possibility of a release of radioactive material. There was no affect on cask confinement or structural and shielding functions of the cask. The ability of the cask and other structures, systems and components important to safety to perform their intended safety functions remains unaffected.

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End of 10CFR 72.48 Report

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Exelon Nuclear
Peach Bottom Atomic Power Station
Units 1, 2 and 3
Docket Nos. 50-171, 50-277, and 50-278

COMMITMENT REVISION REPORT
JANUARY 1, 2012 THROUGH DECEMBER 31, 2012
CHANGE SUMMARIES

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Letter Source: Letter to NRC dated 4/24/91 Regarding Station Blackout Commitments

Exelon Tracking No.: T03634

Nature of Commitment: Demonstrate Emergency Diesel Generator (EDG) operability prior to arrival of a hurricane

Summary of Justification:

The commitment is revised to allow for ensuring EDG operability prior to the onset of a hurricane by administrative means rather than operating the EDGs. Verification of EDG operability can be performed by administrative checks by examining logs or other information to determine if the EDGs are out of service for maintenance or for other reasons. This check would also include a routine inspection. This change is acceptable since it is a well established industry fact that the outcome of surveillances is typically that operability is confirmed. Due to the high quality of the EDGs and normal testing, reasonable assurance exists that the EDGs would be operable.

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Letter Source: Letter to NRC dated 9/28/79, Response to NRC Bulletin 79-19

Exelon Tracking No.: T04456

Nature of Commitment: Develop and implement training for personnel involved in the transfer, packaging and transport of radioactive material

Summary of Justification:

Upgrades in the training program have resulted in substantial improvements in ensuring appropriate training is administered to personnel involved with radioactive material. A standard training program for radioactive material shipping is in place. Therefore, this commitment is considered to be historical in nature. The corrective actions taken were effective and the station is in compliance with requirements. There is no longer a need to track this commitment.

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Letter Source:	Letter to NRC dated 10/31/91, Response to NRC Inspection Report 91-28/28
Exelon Tracking No.:	T01596
Nature of Commitment:	Augmented procedure controls to ensure appropriate monitoring of radioactive material during system breaches

Summary of Justification:

This is a historical commitment. The corrective actions taken were effective and the station is in compliance with NRC requirements. Improved standards and practices within the radiation protection program ensure proper surveys and reviews. The radiation protection standardized process has substantially improved since 1991. There is no longer a need to track this commitment.

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Letter Source:	Letter to NRC dated 5/16/97 involving NRC Notice of Violation for an Unmonitored Release
Exelon Tracking No.:	T04001
Nature of Commitment:	Revise procedure to add guidance concerning potential unmonitored release paths

Summary of Justification:

This is a historical commitment. The corrective actions taken were effective and the station is in compliance with NRC requirements. Improved standards and practices within the radiation protection procedures ensure proper radioactive material control. The radiation protection standardized process has substantially improved since 1997. There is no longer a need to track this commitment.

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Letter Source:	Letter to NRC dated 12/04/84 in response to NRC Inspection Report 84-25/21 involving control of dosimetry
Exelon Tracking No.:	T03188
Nature of Commitment:	Develop a routine test to ensure that lost dosimeters are evaluated

Summary of Justification:

This is a historical commitment. The corrective actions taken were effective and the station is in compliance with NRC requirements. Improved standards and practices within the radiation protection / dosimetry program have addressed this issue and are proceduralized within the Exelon standard documents. Based on the historical nature of this commitment and upgraded standardized practices, there is no longer a need to track this commitment.

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Letter Source:	Letter to NRC dated 5/31/05 involving a response to Security B.5.b requirements
Exelon Tracking No.:	T04579
Nature of Commitment:	Ensure a staging area is established for response equipment for security design basis events

Summary of Justification:

This commitment is being revised to change the staging area location. The new location is an improvement and will enable local law enforcement the ability to dispatch and provide escort of off-site responding emergency apparatus to the site. This improvement was identified as the result of site reviews and drills.

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Letter Source:	Letter to NRC dated 1/7/02 involving the License Renewal application to the NRC
Exelon Tracking No.:	T04340, T04326, T04339
Nature of Commitment:	Routine Inspection of High Pressure Coolant Injection (HPCI) system and Reactor Core Isolation Cooling (RCIC) system turbine casing and lube oil coolers, HPCI lube oil storage tank, HPCI lube oil system flexible hoses and the HPCI gland seal condenser

Summary of Justification:

This commitment is being revised to change the frequency of these routine inspections from 8 to 10 years. Based on significant operating experience, it has been determined that a 10 year inspection frequency is sufficient for proactive identification of system component degradation. This change is based on industry and site experience. It was concluded that this change satisfies the intent of the license renewal commitment.

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Letter Source:	NRC Safety Evaluation Report (SER) dated 9/25/06 involving on Joint Owenr's Group (JOG) Motor-Operated valve (MOV) Periodic Verification Program (NRC Generic Letter 96-05)
Exelon Tracking No.:	T04783
Nature of Commitment:	Implement the JOG MOV verification program in accordance with MPR-2524-A for NRC Generic Letter 96-05 MOVs not in the scope of JOG MOV program final report (i.e., JOG Class D MOVs)

Summary of Justification:

This commitment is being temporarily revised since two MOVs (MO-3-12-018 and MO-2-12-015) were not able to meet the 6-year implementation schedule to be included in the long-term MOV periodic verification program following issuance of the 9/25/06 NRC SER. These two valves are classified as JOG Class D valves due to the use of non-JOG tested disk to guide material pairings in service temperatures greater than 120^{O}F. Operability Evaluations were performed for the two valves that are not yet modified to justify operability until appropriate modifications are completed. These valves are normally open Primary Containment Isolation Valves for the Reactor Water Cleanup System.

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End of Commitment Revision Report

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